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W3F1-2004-0101

October 19, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement 4 to Amendment Request NPF-38-256,
Alternate Source Term
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated July 15, 2004, "License Amendment Request NPF-38-256, Alternate Source Term"
2. Entergy Letter dated August 19, 2004, "License Amendment Request NPF-38-256, Supplement to Alternate Source Term Submittal"
3. Entergy Letter dated September 1, 2004, "Supplement 2 to Amendment Request NPF-38-256, Alternate Source Term"
4. Entergy Letter dated October 13, 2004, "Supplement 3 to Amendment Request NPF-38-256, Alternate Source Term"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) licensing basis to implement an Alternate Source Term (AST) for calculating accident offsite doses and doses to control room personnel as permitted by 10 CFR 50.67. Entergy supplemented this request via References 2, 3, and 4.

In Reference 4, Entergy committed to submit a revised AST dose analysis for the steam generator tube rupture (SGTR) event to fully account for early releases from the affected SG using both atmospheric dump valves (ADVs) for the early rapid cooldown prior to isolation. Also, in Reference 4, Entergy committed to submit a revised AST dose analysis for the large break loss of coolant accident (LBLOCA) to not credit fission product cleanup of elemental iodine due to containment spray. The results of the revised SGTR and LBLOCA dose analysis are provided in the attachment to this letter and supersede the SGTR and LBLOCA dose analysis information previously submitted in References 1 and 3.

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There are no new commitments in this letter. If you have any questions or require additional information, please contact Jerry Burford at 601-368-5755.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 19, 2004.

Sincerely,

FGB/DBM/cbh

Attachment: Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology

cc: (See Next Page)

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Attachment 1

W3F1-2004-0101

Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology

Licensing Report for the Radiological Consequences of Accidents for the Waterford Steam Electric Station, Unit 3 Using Alternative Source Term Methodology

October 14, 2004

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1.0 RADIOLOGICAL CONSEQUENCES UTILIZING NUREG-1465 SOURCE TERMS

1.1. Introduction

Reference 1 submitted a license amendment request to implement an Alternate Source Term (AST) as permitted by 10CFR50.67 for calculating accident offsite doses and doses to control room personnel for Waterford 3. That submittal provided dose consequence analyses for events expected to be limiting, and noted that a second AST submittal, Reference 2, would be made to provide the results of additional analyses. Reference 3 supplemented Reference 1 with Large Break Loss of Coolant Accident (LBLOCA) Shine calculations and an amended LBLOCA offsite and control room dose analysis. Reference 4 provided additional information on the Waterford 3 AST analysis in response to the Reference 5 NRC letter. This submittal provides the following:

- An amendment to Reference 3, Section 5.0, LBLOCA, and
- An amendment to Reference 1, Section 8.0, Steam Generator Tube Rupture (SGTR).

The LBLOCA offsite and control room dose analysis has been revised to eliminate crediting fission product scrubbing of elemental iodine due to containment spray, as discussed in Reference 4. The SGTR analysis has been revised based upon a more conservative interpretation of Regulatory Guide (RG) 1.183 than used in the original analysis of Reference 1. Specifically, the late releases from the affected Steam Generator (SG), in the 6.5 hour to 8.0 hour timeframe when the Atmospheric Dump Valve (ADV) is being used to control level prior to Shutdown Cooling initiation, are now considered in the analysis. The SGTR analysis has also been revised to correct errors in releases which have been documented in the Waterford 3 corrective action program. Specifically, the inputs to the analysis had terminated releases from the affected SG earlier than should have been assumed.

1.2. Common Analysis Inputs and Assumptions

Common analysis inputs and assumptions are described in Section 1.2 of Attachment 2 of Reference 1. Some inputs and assumptions are identified as being specific to specific events evaluated.

1.3. Control Room Air Conditioning System and Control Room Ventilation Model

The control room air conditioning system and control room ventilation model are described in Section 1.3 of Attachment 2 of Reference 1. The description includes event-specific control room unfiltered in-leakage assumptions for the events evaluated in Reference 1 which are not applicable to events evaluated in this submittal. Specific control room ventilation modeling changes for the SGTR reanalysis are discussed in that section of this submittal. The in-leakage assumptions for events evaluated in this submittal are:

Sequence Type	Control Room Unfiltered In-leakage Modeled
LBLOCA	100 CFM
SGTR	100 CFM

1.4. Exceptions to Regulatory Guide (RG) 1.183

Exceptions applicable to this submittal are identified in Section 1.4 of Attachment 2 of References 1 and 3.

2.0 CONCLUSIONS

A summary of the calculated dose consequences of all events is presented in Table 2-1. The events meet the acceptance criteria for the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Main Control Room (MCR). SGTR results are presented for the Pre-existing Iodine Spike (PIS) and Event Generated Iodine Spike (GIS) cases.

TABLE 2-1
SUMMARY OF RESULTS

Event Scenario	Dose Consequences			Acceptance Criterion
	EAB	LPZ	MCR	EAB&LPZ/MCR
LBLOCA	5.295	2.369	1.466	25/5
SGTR – PIS	0.987	0.213	4.85	25/5
SGTR – GIS	0.435	0.088	2.56	2.5/5

Notes: All Results are presented in units of rem Total Effective Dose Equivalent (TEDE).

Detailed discussions for each individual event are presented in Sections 4 and 5. The detailed analyses for each event demonstrate that radiological consequences meet the TEDE dose acceptance limits for off-site dose. The radiological consequences for MCR dose for all events are ≤ 5 Rem TEDE.

The total dose to control room personnel from the LBLOCA inhalation dose and doses from various post-LBLOCA shine sources is also ≤ 5 Rem TEDE. This is presented in more detail in Section 6 of Reference 3.

3.0 REFERENCES

1. W3F1-2004-0053, "License Amendment Request NPF-38-256, Alternate Source Term, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," July 15, 2004.

2. W3F1-2004-0071, "License Amendment Request NPF-38-256, Alternate Source Term, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," August 19, 2004

3. W3F1-2004-0073, "License Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," September 1, 2004.

4. W3F1-2004-0095, "Supplement 3 to Amendment Request NPF-38-256, Alternate Source Term, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," October 13, 2004.

5. Letter from N. Kalyanam (NRC) to J. Venable (Entergy), "Waterford Steam Electric Station, Unit 3 (Waterford 3) – Request for Additional Information Related to Revision to Facility Operating License and Technical Specifications – Extended Power Uprate Request (TAC NO. 1355) and Alternate Source Term Request (MC3789)," September 29, 2004.

6. W3F1-2003-0074, "License Amendment Request NPF-38-249, Extended Power Uprate, Waterford Steam Electric Station, Unit 3, Docket No. 50-382, License No. NPF-38," November 13, 2003.

4.0 LARGE BREAK LOSS OF COOLANT ACCIDENT (LBLOCA)

The LBLOCA dose analysis has been revised to eliminate crediting fission product scrubbing of elemental iodine due to containment spray.

The design basis LBLOCA is postulated as a break in the reactor coolant pressure boundary piping. An abrupt failure of the main reactor coolant piping is assumed to occur and it is assumed that the Emergency Core Cooling System (ECCS) fails to prevent the core from experiencing significant degradation. This is considered a Limiting Fault event. Activity from the core is released to containment and subsequently to the environment by means of containment leakage or leakage from the ECCS. Release of core radioactive inventory to the containment is postulated in accordance with RG 1.183 guidance on activity release and timing for the gap fraction release and early-in vessel release phases.

Other than adoption of the RG 1.183 methodology, the LBLOCA dose analysis is relatively unchanged compared to the analysis presented in Extended Power Uprate (EPU) Licensing Amendment Request, Reference 6.

4.1. Input Parameters and Assumptions

The input parameters and assumptions are listed in Table 4-1. Certain assumptions are discussed in additional detail below.

4.1.1. Source Term

Table 1-1 of Reference 1 documents the core inventory assumed for the LBLOCA radiological dose calculations. Two separate ORIGEN calculations were conducted for the Waterford 3 EPU project to provide core inventories. One calculation was performed to determine the gap fission product activities in peak power rods. A second calculation was performed to determine the core-wide fission product inventory. There was generally good agreement between these two calculations, with their slightly separate biases. A LOCA source term (Table 1-1 of Reference 1) was constructed using the more conservative (larger) value of core inventory from the two sources. Several isotopes are modeled in RADTRAD for which inventories were not calculated in the ORIGEN calculations. For those isotopes, the default Pressurized Water Reactor (PWR) core inventories (on a Ci/MWt basis) from NUREG/CR-6604 were assumed.

The release fractions applied to the various species of fission products are consistent with Table 2 of RG 1.183 for PWR core inventory fraction releases for the gap release phase and early in-vessel phase of release. Timing of the release phases is from Table 4 of RG 1.183 for LOCA release phase timing. This information is documented in Table 4-2.

The reactor coolant initial activity is insignificant in comparison with the releases due to the postulated core damage for this event.

4.1.2. Iodine Chemical Form

As listed in Table 4-2, iodine released to containment is assumed to be 95% aerosol/particulate, 4.85% elemental, and 0.15% organic. This is consistent with Section 3.5 of RG 1.183.

The radioiodine postulated to be available for release to the environment through Engineered Safety Feature (ESF) leakage is assumed to be 97% elemental and 3% organic. This is consistent with Section 5.6 of RG 1.183 Appendix A.

4.1.3. Release Pathways

Activity from the reactor coolant system and the failed core is released into the containment. Releases are postulated from the containment to the environment by three containment air leakage pathways (Reactor Auxiliary Building (RAB)/Controlled Ventilation Areas System (CVAS), Shield Building, and Direct Bypass) and by leakage from ESF systems (safety injection and containment spray) which take suction, upon recirculation, from the safety injection sump. The fraction of the release associated with each of the three containment air leakage pathways is specified in Table 1-2 of Reference 1.

The containment is modeled as a sprayed and an unsprayed region, where the sprayed region is subject to fission product removal due to the action of the containment sprays (80% of the containment volume is assumed subject to containment spray). Consistent with RG 1.183, Appendix A, a mixing rate due to natural convection between the sprayed and unsprayed regions of containment can be assumed to equal two turnovers of the unsprayed region per hour; this assumption has been adopted for the LBLOCA dose calculation. This is considered a conservative assumption since at least one containment fan cooler is assumed available, providing forced circulation mixing within the containment.

The containment is assumed to leak at the design rate of 0.50 w/o per day for the first 24 hours, and at half that rate (0.25 w/o per day) thereafter. This is consistent with RG 1.183, Appendix A.

Direct bypass releases are assumed to be released unfiltered directly to the environment. Releases to the area of the RAB serviced by the CVAS are assumed to be filtered and directly released to the environment; no credit is taken for holdup in the RAB. Shield Building holdup and dilution is modeled. A Shield Building Ventilation System (SBVS) maximum flow rate of 11,000 CFM per train is modeled. It is assumed that when one train is operating, flow is induced in the second train, which is assumed to be unfiltered. The Shield Building pressure transient following a LBLOCA is documented in UFSAR Figures 6.2-47a and 6.2-47b. Conservatively, when the SBVS is in exhaust mode releasing to the environment, a total flow of 24,244 CFM is assumed with an 89.8% filter efficiency; this very conservatively assumes that even though each train is operating, it is also inducing the unfiltered flow. When the SBVS is in recirculation, only a nominal flow rate of 10,000 CFM is assumed and it is assumed that only one train is operating. Thus, the modeling of the SBVS is very conservative. A small effective exhaust flow of approximately 35 CFM is assumed for long-term operation of the SBVS (i.e., beyond about 43 hours); the remaining flow, based on the nominal 10,000 CFM flow rate, is assumed to be in recirculation. After 168 hours, the SBVS is assumed to be exhausting to account for the postulated failure of the containment maintenance hatch seal.

The analysis considers a leak rate of 0.5 GPM from ECCS systems that are recirculated and may leak to locations serviced by the CVAS system in the RAB. While no credit is taken for holdup and dilution in the RAB, CVAS filtration is credited. A flashing fraction of 10% is assumed, consistent with RG 1.183. The release is assumed to begin at the postulated earliest time before ECCS recirculation of 23.4 minutes.

4.1.4. Removal Coefficients

Containment spray removal coefficients consistent with NUREG-0800, Section 6.5.2 are assumed. One train of Containment Spray is assumed to operate following a LOCA, with a minimum flow rate of 1750 GPM. These values are documented in Table 4-1. Removal of elemental iodine from containment atmosphere is not modeled for the purpose of determining offsite dose or dose due to radionuclides entering the control room. For determination of control room shine doses, a maximum Partition Factor (PF) of 200 has been assumed; this occurs at 1.8 hours. The removal coefficient for particulate/aerosol iodine is assumed, consistent with NUREG-0800, to decrease by a factor of ten when the airborne inventory has dropped to 2% of the total particulate iodine released to the containment (a PF of 50). This also occurs after 1.8 hours. The responses to Questions 23 and 24 in Reference 4 provide additional detail regarding the modeling of removal of fission products by containment spray for LBLOCA.

Per RG 1.183 Appendix A Section 3.2, reduction of airborne activity by natural deposition within the containment may be credited for LOCA. The Powers 10% Aerosol Deposition is specified for natural deposition of aerosols/particulates. This model is described in NUREG/CR-6604. The lower bound of this deposition model (10th percentile) is specified. Use of this model is consistent with RG 1.183, Appendix A, Section 3. The guidance of NUREG-0800, Section 6.5.2 is applied for natural deposition of elemental iodine. Natural deposition removal coefficients are documented in Table 4-1.

4.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The LBLOCA dose model assumes an unfiltered in-leakage of 100 CFM for the event duration. It is assumed that the preferred control room intake is selected at two hours into the event, at which time the operators also initiate the pressurized mode of control room operation. However, no credit is taken in this event scenario for the lower in-leakage during the pressurized mode of operation.

4.2. Results

The radiological consequence results in Rem TEDE are listed below and compared with the acceptance criteria for LOCA provided by RG 1.183 and 10CFR50.67:

	LBLOCA	Acceptance Criteria
EAB (worst two hour dose)	5.295	25 Rem TEDE
LPZ (worst 30 day duration)	2.369	25 Rem TEDE
MCR	1.466	5 Rem TEDE

Thus, the radiological consequences for LBLOCA are < 25 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR, based on a maximum control room unfiltered in-leakage of 100 CFM.

TABLE 4-1
ASSUMPTIONS USED FOR LBLOCA RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
Containment Leak Rate:	0.50 % volume/day (0-24 hours) 0.25 % volume/day (24 hours - 30 days)
Natural Deposition:	
Elemental	0.40/hr
Organic	0
Particulate	Powers 10% Aerosol Decontamination Factor
Spray Fission Product Removal (LBLOCA):	
Elemental	0**
Organic	0
Particulate	3.596/hr (until PF = 50) 0.3596/hr (once PF > 50) ** Shine dose calculations assume 20/hr with a maximum PF of 200.
Containment Mixing Rate Between Sprayed and Unsprayed Regions:	17,122 CFM
Maximum Spray Delay Time:	60 seconds
Containment Leakage Pathway:	
Controlled Ventilation Area System (CVAS) Filtration (Reactor Auxiliary Building)	54%
Shield Building	40%
Unfiltered Direct Bypass	6%
Control Room Parameters	See Table 1-2 of Reference 1

Main Control Room X/Q Assumed:

Time	Unfiltered In-leakage	Pressurization Flow
0-2 hr	2.77E-03	2.77E-03
2-8 hr	1.78E-03	3.90E-04*
8-24 hr	7.22E-04	1.79E-04*
1-4 days	5.27E-04	1.37E-04*
4-30 days	4.05E-04	1.08E-04*

* factor of 4 reduction credited per SRP 6.4.

TABLE 4-2
SOURCE TERM ASSUMPTIONS: LBLOCA RADIOLOGICAL ANALYSIS

Core Inventory Fraction Released into Containment:

Group	Gap Release Phase	Early In-Vessel Phase
Noble Gas	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metal	0.00	0.0025
Cerium group	0.00	0.0005
Lanthanides	0.00	0.0002

LOCA Release Phases:

Phase	Onset	Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

Iodine Chemical Form (release to containment):

Aerosol/Particulate	95%
Elemental	4.85%
Organic	0.15%

Iodine Chemical Form (ESF system leakage):

Elemental	97%
Organic	3%

Time (sec.)	Event
0.0	SG tube rupture occurs
442.7	Core protection calculator hot leg saturation trip condition reached
445.4	Trip breakers open
449	Loss of off-site power
450	SG ADVs open
485	Safety injection actuation signal
>875	Operator takes manual control of SG ADVs. Plant cooldown initiated by steaming using ADV of both SGs.
1980	Operator isolates affected SG, terminating cooldown release from ADV of affected SG.
23630	Operator opens ADV to the affected SG as needed to maintain level below 94% wide range.
28800	Shutdown cooling entry conditions achieved; release stopped.

As described in EPU Licensing Amendment Request, Section 2.13.6.3, the SGTR analysis assumes that a LOOP occurs 3 seconds after reactor trip. This is consistent with the assumption made on other plants and included in CESSAR FSAR Chapter 15. On the basis that no fuel failure occurs for the event, the radiological analyses would be insensitive to this assumption for the Waterford 3 design.

As described in EPU Licensing Amendment Request, Section 2.13.6.3.2, during the SGTR, a total of 325,700 lbm of primary coolant passes through the rupture into the affected SG. Prior to reactor trip, both SGs are steaming normally to the condenser. Due to the high PFs and to the release geometry, releases from this source do not contribute to MCR dose. Following reactor trip, both SGs are steamed through the ADVs. The operator takes manual control of the ADVs at approximately 875 seconds and uses the ADVs on both SGs to cooldown the plant; at 1980 seconds the affected SG is isolated. 139,000 lbm of steam is released from the affected SG between trip and 1980 seconds. Note that although the EPU Licensing Amendment Request documents operator action to open the ADV to the affected SG to maintain level below 94%, other operator actions are available in the EOPs to maintain SG level and prevent SG overfill. These late releases, between 6.5 and 8.0 hours in the event scenario, are included in the analysis. Thus, all releases from both generators until shutdown cooling is entered are considered. There are no subsequent releases from shutdown cooling entry until achieving cold shutdown. A total of 245,600 lbm of steam is released through the ADV of the affected SG during the cooldown.

The majority of the cooldown of the plant is performed by steaming the unaffected SG. A total of 910,100 lbm of steam is released through the unaffected generator's ADV during the plant cooldown. Radioactivity release through the intact SG is assumed due to primary-to-secondary SG tube leakage of 0.375 gpm per generator (540 gpd). Once shutdown cooling is initiated, there are no further releases postulated from the SGs.

5.1.4. Removal Coefficients

Iodine releases to the affected SG are assumed to flash to vapor whenever the top of the SG U-tubes are uncovered and are available for release without mitigation. The flashing fraction is based on the difference between the primary side fluid enthalpy and the saturation enthalpy on the secondary side.

An iodine PF of 100 is assumed for activity transported to the secondary side prior to reactor trip. The un-flashed portion of the tube rupture flow mixes with the SG inventory and is released with a PF of 100. RG 1.183, Appendix F, for PWR SGTR, endorses the Appendix E, Position 5.5.4 which calls for assuming an iodine PF of 100.

A PF of 100 is assumed for the 0.375 gpm primary-to-secondary leak rate assumed for the unaffected SG

Prior to reactor trip, any releases from the condenser could be assumed to have an iodine PF of 100 applied to those releases. Credit for additional iodine removal in the condenser is not assumed. The pre-trip releases from the condenser are assumed to contribute only to off-site dose as discussed in Section 5.1.5 below.

All noble gas release to the secondary side is assumed to be immediately released to the environment.

5.1.5. Main Control Room Model

The MCR ventilation model is described in Section 1.3 of Reference 1. The SGTR dose model for secondary steaming assumes an unfiltered in-leakage of 100 CFM. It is assumed that the preferred control room intake is selected when pressurization flow is established.

The analyses assume a maximum control room filtered intake flow of 225 CFM and a minimum control room filtered intake flow of 0 CFM for the duration of the event. These are mutually exclusive but bounding assumptions.

Due to geometry considerations, the pre-trip releases from the condenser are assumed to contribute only to the off-site dose consequences. Because the condenser release point and the worst case ADV release locations are in the opposite directions from the worst case control room air intake, and since the MCR envelope is isolated on any high radiation signal prior to the radiation entering the envelope, releases from the condenser are not assumed to contribute to the control room dose. Were wind speed and direction conditions such that releases from the condenser were to be directed to the MCR air intakes, the atmospheric dispersion factors for the ADVs would be greatly reduced. Also, the control room would be isolated on a high radiation signal prior to any of the release entering the control room envelope. Thus, any scenario involving releases from the condenser to the MCR would be less limiting than scenarios involving worst case atmospheric dispersion factors for releases from the ADVs to the control room.

Because the control room envelope would be isolated on a high radiation signal prior to any of the release reaching the envelope, the main control room ventilation system is assumed to be in either of its radiological emergency modes (pressurized or recirculation) once the ADVs open for the subsequent duration of the event.

At 450 seconds, the ADVs (and MSSVs) first open following the plant scram. It is assumed that all releases from the main steam lines are through the ADVs, which have worse (higher) χ/Q values than the MSSVs. The 5% probability level χ/Q values for the ADVs to the two control room intakes are:

MCR Atmospheric Dispersion Factors, χ/Q (s/m^3)

Time	East ADV to East MCR Air Intake	East ADV to West MCR Air Intake	West ADV to East MCR Air Intake	West ADV to West MCR Air Intake
450 s - 2 hrs	1.06E-01	1.23E-03	1.36E-03	7.50E-03
2-8 hrs	7.45E-02	8.31E-04	8.29E-04	5.62E-03

The unfiltered in-leakage to the control room will be conservatively assumed to be subject to the worst case χ/Q values, i.e., those for East ADV releases to East MCR Air Intake. Since about 98% of the releases are from SG #1, that is assumed to be the East SG, and it is assumed to be the source for all of the activity for the unfiltered in-leakage. This χ/Q for the East ADV to the East MCR intake is applied to the entire unfiltered in-leakage for the 8 hour period for which a release is postulated.

The Waterford 3 control room will be placed in recirculation mode automatically upon either a high radiation signal or a Safety Injection Actuation Signal (SIAS). Operator action is required to pressurize the control room. Note that per NUREG-0800, Section 6.4, operator action to switch the assumed location of the control room emergency air intake to the more favorable location may be assumed. Because the dominant unfiltered inleakage term assumes the less favorable MCR air intake location, per NUREG-0800, a χ/Q corresponding to the χ/Q for the more favorable intake, divided by a factor of 4, may be assumed for the pressurization flow. Operators would diagnose which SG has been subject to the tube rupture and use the MCR Air Intake least impacted by releases from the ADV of the affected SG. Thus, since Operator action is required to establish pressurization flow, a reduced χ/Q may be applied to the pressurization flow.

MCR Atmospheric Dispersion Factors, χ/Q (s/m^3)
After Operator Action to Select Preferred Air Intake

Time	East ADV to West MCR Air Intake	West ADV to West MCR Air Intake
0-2 hrs	3.075E-04	7.50E-03
2-8 hrs	2.0775E-04	5.62E-03

Since each SG contributes to the source of the pressurization flow, a scaled χ/Q can be developed to account for the relative contribution from each of the modeled sources. Thus, for each of the three dose cases (PIS, GIS, and Noble Gas), a scaled effective χ/Q can be defined as:

$$\chi/Q_{eff} = ((R_1 \times \chi/Q_1) + (R_2 \times \chi/Q_2))/(R_1 + R_2)$$

where R_i is the release fraction for each source/volume (i.e., SG_1 or SG_2) and χ/Q_i is the corresponding atmospheric dispersion factor. The R_i values are based directly on the curie releases from the CENTS analyses documented in Section 2.13.6.3.2 of the EPU Licensing Amendment Request.

The effective control room χ/Qs for the pressurization flow for each case are computed below:

χ/Q (s/m^3):	0-120 min	2-8 hr
SG_1 to MCR	0.000308	0.00020775
SG_2 to MCR	0.0075	0.00562
Release Fractions:		
SG_1, PIS	0.37030	0.26687
SG_1, GIS	0.10524	0.84911
SG_1, NG	0.35556	0.58939
SG_2, PIS	0.00218	0.00638
SG_2, GIS	0.00065	0.00970
SG_2, NG	0.00134	0.00280
Effective χ/Q Values (Filtered In-leakage):		
PIS	0.000350	0.000334
GIS	0.000352	0.000269
NG	0.000334	0.000233

5.2. Results

The radiological consequence results in Rem TEDE are listed below and compared with the acceptance criteria for SGTR provided by RG 1.183, Table 6 and 10CFR50.67:

	TEDE Dose	Acceptance Criteria
PIS case:		
EAB (worst two hour dose)	0.987	25 Rem TEDE
LPZ (duration)	0.213	25 Rem TEDE
Main Control Room	4.85	5 Rem TEDE
GIS case:		
EAB (worst two hour dose)	0.435	2.5 Rem TEDE
LPZ (duration)	0.088	2.5 Rem TEDE
Main Control Room	2.56	5 Rem TEDE

Thus, the radiological consequences for SGTR are < 25 Rem TEDE for the EAB and LPZ doses and < 5 Rem TEDE for the MCR for the PIS case. Radiological consequences are < 2.5 Rem TEDE for EAB and LPZ doses for the GIS case. These are based on a 540 gpd primary-to-secondary leak rate for the unaffected SG and maximum control room unfiltered in-leakage of 100 CFM.

Activity releases for the SGTR are as follows:

	DEI-131 Release		Noble Gas Release (Table 1-4 isotopic distribution)	
	2 hr EAB (Ci)	8 hr LPZ (Ci)	2 hr EAB (Ci)	8 hr LPZ (Ci)
PIS	133.22	183.31	28,516	69925.5
GIS	7.24	51.31	28,516	69925.5

The portion of the 2 hour releases through the condenser prior to the ADVs opening are 64.94 Ci for the PIS Iodine release, 1.81 Ci for the GIS Iodine release, and 3559 Ci Noble Gas.

TABLE 5-1
ASSUMPTIONS USED FOR SGTR RADIOLOGICAL ANALYSIS

Core Power Level:	3735 MWt
RCS Noble Gas Activity:	See Table 1-4 of Reference 1
Core Inventory:	See Table 1-1 of Reference 1
RCS Initial Activity:	100/Ē μCi/gm
Pre-existing Iodine Spike (PIS):	60 μCi/gm DEI-131
Accident Generated Iodine Spike (GIS):	1.0 μCi/gm DEI-131
Iodine Spiking Factor:	335
Secondary Coolant Initial Activity:	0.1 μCi/gm DEI-131
Fraction of Fuel Rods in Core Failing:	0%
Iodine Chemical Form:	
Elemental	97%
Organic	3%
Particulate	0%
Primary-to-Secondary Leak Rate (unaffected SG):	540 gpd
Steaming PF:	100
Steam Releases:	
Affected SG, time of reactor trip to isolation (1980 sec)	139,000 lbm
Affected SG, time of reactor trip to 8 hours	245,600 lbm
Intact SG, time of reactor trip to 2 hours	351,400 lbm
Intact SG, time of reactor trip to 8 hours	910,100 lbm
Duration of Release:	8.0 hours
Control Room Parameters	See Table 1-2 of Reference 1
Main Control Room χ/Q assumed:	See Section 5.1.5
Pressurization Flow:	225 CFM (maximum) 0 CFM (minimum, 0-8 hours)
Unfiltered In-leakage	100 CFM